



UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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February 14, 2007

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Fort Calhoun Station FC-2-4 Adm.
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SUBJECT: FORT CALHOUN STATION - NRC INTEGRATED INSPECTION
REPORT 05000285/2006005

Dear Mr. Ridenoure:

On December 31, 2006, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Fort Calhoun Station. The enclosed integrated inspection report documents the inspection findings, which were discussed on January 11, 2006, with Mr. Jeff Reinhart, Site Director, and other members of your staff.

The inspections examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

This report documents two NRC-identified and five self-revealing findings of very low safety significance (Green). All of these findings were determined to involve violations of NRC requirements. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these findings as noncited violations (NCV), consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest the violations or significance of the NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011-4005; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Fort Calhoun Station facility.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, and its enclosure, will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at <http://www.nrc.gov/reading-rm/adams.html> (the Public Electronic Reading Room).

Sincerely,

/RA/

Jeff A. Clark, Chief
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Docket: 50-285
License: DPR-40

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NRC Inspection Report 05000285/2006005
w/Attachment: Supplemental Information

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RIV:RI:DRP/E	SRI:DRP/E	C:DRS/EB1	C:DRS/OB	C:DRS/PSB
LMWilloughby	JDHanna	WBJones	ATGody	MPShannon
T-JAC	T-JAC	/RA/	/RA/	/RA/
2/5/07	2/5/07	2/5/07	2/7/07	2/8/07
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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket: 50-285

License: DPR-40

Report: 05000285/2006005

Licensee: Omaha Public Power District

Facility: Fort Calhoun Station

Location: Fort Calhoun Station FC-2-4 Adm.
P.O. Box 399, Highway 75 - North of Fort Calhoun
Fort Calhoun, Nebraska

Dates: October 1 through December 31, 2006

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TABLE OF CONTENTS

SUMMARY OF FINDINGS	-3-
REPORT DETAILS	-7-
1. REACTOR SAFETY	-7-
1R01 <u>Adverse Weather Protection (71111.01)</u>	-7-
1R04 <u>Equipment Alignments (71111.04)</u>	-8-
1R05 <u>Fire Protection (71111.05)</u>	-8-
1R08 <u>Inservice Inspection Activities (71111.08)</u>	-9-
1R11 <u>Licensed Operator Regualification Program (71111.11)</u>	-11-
1R12 <u>Maintenance Effectiveness (71111.12)</u>	-11-
1R13 <u>Maintenance Risk Assessments and Emergent Work Control (71111.13)</u>	-12-
1R15 <u>Operability Evaluations (71111.15)</u>	-12-
1R19 <u>Postmaintenance Testing (71111.19)</u>	-17-
1R20 <u>Refueling and Other Outage Activities (71111.20)</u>	-18-
1R22 <u>Surveillance Testing (71111.22)</u>	-20-
1EP4 <u>Emergency Action Level and Emergency Plan Changes (71114.04)</u>	-20-
2. RADIATION SAFETY	-21-
2OS1 <u>Access Control to Radiologically Significant Areas (71121.01)</u>	-21-
2OS2 <u>ALARA Planning and Controls (71121.02)</u>	-24-
4. OTHER ACTIVITIES	-27-
4OA1 <u>Performance Indicator Verification (71151)</u>	-27-
4OA2 <u>Identification and Resolution of Problems (71152)</u>	-28-
4OA3 <u>Event Followup (71153)</u>	-28-
4OA5 <u>Other Activities</u>	-31-
4OA6 <u>Meetings</u>	-31-
SUPPLEMENTAL INFORMATION	A-1
KEY POINTS OF CONTACT	A-1
LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED	A-1
LIST OF DOCUMENTS REVIEWED	A-2
LIST OF ACRONYMS	A-10

SUMMARY OF FINDINGS

IR 05000285/2006005; 10/01/2006 - 12/31/2006; Fort Calhoun Station, Integrated Resident and Regional Report; Operability Evaluations, Refueling and Other Outage Activities, Access Control to Radiologically Significant Areas, ALARA Planning and Controls, Event Follow-up.

The report covered a 3-month period of inspection by a senior resident inspector, a resident inspector and announced inspections by a reactor inspector, a senior project engineer, a project engineer, a senior emergency preparedness inspector and a health physicist. Seven Green findings, all of which were noncited violations, were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, "Significance Determination Process." Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified Findings and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a Green noncited violation of 10 CFR 50, Appendix B, Criterion XVI for the licensee's failure to promptly identify and correct a degraded component cooling water pump. The failure to recognize and fix this condition led to the pump being more likely to fail upon a valid demand to start.

This finding was determined to be greater than minor because the condition had an impact on availability/reliability of the component and thus affected the "Equipment Performance" attribute under the Mitigating Systems cornerstone. The inspectors evaluated this finding using Manual Chapter 0609, Appendix A, and determined that it was of very low safety significance (Green). This conclusion was reached because the finding was not a design or qualification deficiency, the finding did not represent a loss of safety function, was not an actual loss of safety function of a single train for greater than its Technical Specification Allowed Outage time, did not represent an actual loss of safety function for non-Technical Specification equipment, and was not potentially significant due to external events such as flooding, seismic occurrences, etc. This violation was entered into the licensee's corrective action program as Condition Report 200603835. This finding has a crosscutting aspect in the area of problem identification and resolution because the licensee failed to identify and correct the condition despite numerous opportunities to do so (Section 1R15.b.1).

- Green. A Green self-revealing finding was identified for failure of operators to follow a standing operational procedure as required by Technical Specification 5.8.1.a. This failure resulted in less than the minimum number of raw water pumps required for decay heat removal from the spent fuel pool.

This finding was determined to be greater than minor in that it affected the "Configuration Control" attribute of the Mitigating Systems cornerstone, specifically "Shutdown Equipment Alignment." The inspectors attempted to use Manual Chapter 0609, Appendix G, because the condition occurred during shutdown conditions. The inspectors were unable to do so because an assumption contained in the worksheets was that fuel was in the reactor vessel. During this transient, all fuel was located in the spent fuel pool. Regional management determined that the finding was of very low safety significance (Green). The finding was evaluated considering Manual Chapter 0609, Appendix G, as a bounding case and was used as guidance to determine the significance of the finding. This violation was entered into the licensee's corrective action program as Condition Report 200604505. This finding has a crosscutting aspect in the area of human performance associated with work practices because the operator failed to use error prevention techniques like self-checking and peer checking, which would have prevented this event (Section 1R20).

- Green. A Green self-revealing finding was identified for failure to follow Technical Specification 5.8.1a required procedures during testing. This condition resulted in the damage to safety-related equipment and potential over-pressurization of chemical and volume control system and high-pressure safety injection piping.

This finding was determined to be greater than minor in that it affected the "Configuration Control" attribute of the Mitigating Systems cornerstone, specifically "Shutdown Equipment Alignment." The inspectors evaluated this finding using Manual Chapter 0609, Appendix G, because the condition occurred during shutdown conditions. Using Checklist 2, the inspectors determined that the finding screened as Green because the condition did not increase the likelihood that a loss of decay heat removal would occur. This violation was entered into the licensee's corrective action program as Condition Report 200605430. This finding has a crosscutting aspect in the area of human performance associated with work practices because the operator failed to use error prevention techniques like self-checking and peer checking, which would have prevented this event (Section 4OA3.3).

Cornerstone: Barrier Integrity

- Green. The inspectors identified a noncited violation of Technical Specification 2.4. The violation was identified as a result of the licensee's failure to complete corrective actions two years ago that caused the licensee to incorrectly determine the operability of component cooling water inlet and outlet valves. These valves supply component cooling water to the containment air-cooling and containment air-cooling and filtering units.

The finding was more than minor since it affected the "Containment Configuration Control" attribute of the Barrier Integrity cornerstone. Using Significance Determination Process, Manual Chapter 0609, the phase one analysis directs the use of Appendix H, since the finding involves the actual reduction in defense-in-depth for the atmospheric pressure control. Manual Chapter 0609 Appendix H,

characterized the finding as having a very low safety significance because it was determined to have no impact on core damage frequency or large early release frequency. The finding also has a crosscutting aspect in the problem identification and resolution area because the licensee failed to take appropriate corrective actions to address the safety issue in a timely manner (Section 1R15.b.2).

Cornerstone: Occupational Radiation Safety

- Green. The inspectors identified a self-revealing, noncited violation of Technical Specification 5.11.1, in which a worker failed to obtain a high radiation area access authorization and associated radiological briefing prior to entering the posted area. Specifically, on October 24, 2006, a worker entered the containment building on a radiation work permit for rigging and equipment moves. This assignment did not require entry into a posted high radiation area. After entering the containment building and beginning work, the individual's foreman reassigned the person to a job in a posted high radiation area. The individual did not change radiation work permits and did not receive the high radiation area briefing prior to starting work in the new area. This issue was entered into the licensee's corrective action program.

This finding is greater than minor because it is associated with one of the cornerstone attributes (exposure/contamination control) and affects the Occupational Radiation Safety cornerstone objective, in that the failure to obtain authorization for entry into the posted high radiation area and the radiological briefing could result in additional personnel exposure. Using the Occupational Radiation Safety Significance Determination Process, the inspectors determined that this finding was of very low safety significance because it did not involve: (1) an as low as is reasonably achievable finding, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess doses. Additionally, this finding has a crosscutting aspect in the area of human performance work control because the foreman failed to appropriately coordinate work activities and evaluate the impact of changes to work assignments (Section 2OS1.1).

- Green. The inspectors identified a self-revealing, noncited violation of Technical Specification 5.11.1.b, in which a contractor's ALARA Coordinator failed to wear an alarming device that could be heard while working in an high radiation area. Specifically, on October 24, 2006, the individual inadvertently signed in on a radiation work permit task that was suspended, and entered an high radiation area inside the containment building. The access control computer automatically set the dosimeter alarms for suspended tasks at one millirem for dose and one millirem/hr for a dose rate. When the individual entered the high radiation area with high background noise levels, the individual was unable to hear the dosimeter alarm after it accumulated one millirem-integrated dose. The individual worked in the area for a total of 1.7-hours. Upon exiting, the individual noticed the dosimeter was alarming and had accumulated a total dose of six-millirem. This issue was entered into the licensee's corrective action program. This finding is greater than minor because it is associated with one of the

cornerstone attributes (exposure/contamination control) and affects the Occupational Radiation Safety cornerstone objective, in that the failure to provide adequate alarming dosimetry resulted in additional personnel exposure. Using the Occupational Radiation Safety Significance Determination Process, the inspectors determined that this finding was of very low safety significance because it did not involve: (1) an as low as reasonable achievable finding, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess doses. Additionally, this finding has a crosscutting aspect in the area of human performance work practices because the worker failed to use error prevention tools such as self- and peer-checking (Section 2OS1.2).

- Green. The inspectors identified a self-revealing, noncited violation of Technical Specification 5.8.1.a, in which instructions for the use of a high-efficiency particulate air filtration units were not adequately incorporated into radiation work permit instructions resulting in the contamination of three workers. Specifically, on September 28, 2006, three individuals received intakes of radioactive material while cutting instrument lines from the bottom of the pressurizer. The work area was set up using scaffolding, with a small work platform, to access the bottom of the pressurizer and an high efficiency particulate air ventilation unit in place on the floor below the work platform with ductwork extending to the work platform. The workers were given a briefing on dosimetry, dress requirements, and dose rates just prior to the start of the job; however, neither the briefing nor the radiation work permit addressed the proper placement of the high efficiency particulate air hose during the cutting evolution. Consequently, the three workers were assigned doses of 60-, 75-, and 86-millirem committed effective dose equivalent respectively. This issue was entered into the licensee's corrective action program.

This finding is greater than minor because it is associated with one of the cornerstone attributes (exposure/contamination control) and affects the Occupational Radiation Safety cornerstone objective, in that the failure to incorporate adequate work instructions in the radiation work permit resulted in additional personnel exposure. Using the Occupational Radiation Safety Significance Determination Process, the inspectors determined that this finding was of very low safety significance because it did not involve: (1) an ALARA finding, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess doses. Additionally, this finding has a crosscutting aspect in the area of human performance resources because the licensee failed to provide complete and accurate work instructions in the radiation work permit (Section 2OS2.2).

B. Licensee-Identified Violations

None

REPORT DETAILS

Summary of Plant Status

The unit began this inspection period in Mode 5 during a refueling outage with all fuel located in the Spent Fuel Pool. On December 3, 2006, the reactor was made critical following completion of the outage. On December 13, 2006, reactor power was increased to 100 percent where the plant remained until the end of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

a. Readiness for Seasonal Susceptibilities

The inspectors completed a review of the licensee's readiness of seasonal susceptibilities involving extreme low temperatures. The inspectors: (1) reviewed plant procedures, the Updated Safety Analysis Report (USAR), and Technical Specifications (TS) to ensure that operator actions defined in adverse weather procedures maintained the readiness of essential systems; (2) walked down portions of the systems listed below to ensure that adverse weather protection features (heat tracing, space heaters, weatherized enclosures, temporary chillers, etc.) were sufficient to support operability, including the ability to perform safe shutdown functions; (3) evaluated operator staffing levels to ensure the licensee could maintain the readiness of essential systems required by plant procedures; and (4) reviewed the corrective action program to determine if the licensee identified and corrected problems related to adverse weather conditions.

- December 6, 2006, supply auxiliary steam to a condensate storage tank, installation of stop logs in circulating water discharge tunnel, and inspection of the heat tracing of auxiliary feedwater supply to the Diesel-Driven Auxiliary Feedwater Pump FW-54

Documents reviewed by the inspectors included: Work Order (WO) 00244139-01, "Install Stop Logs in CW System Discharge Tunnel."

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignments (71111.04)

.1 Partial Equipment Walkdowns

a. Inspection Scope

The inspectors: (1) walked down portions of the two risk-important systems listed below and reviewed plant procedures and documents to verify that critical portions of the selected systems were correctly aligned; and (2) compared deficiencies identified during the walkdowns to the licensee's USAR and Corrective Action Program to ensure problems were being identified and corrected.

- November 29 - December 1, 2006, Safety Injection (SI) System following its release from shutdown cooling operations
- November 24, 2006, Safety-related portions of the Auxiliary Feedwater System.

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Quarterly Fire Inspection Tours

a. Inspection Scope

The inspectors walked down the six plant areas listed below to assess the material condition of active and passive fire protection features and their operational lineup and readiness. The inspectors: (1) verified that transient combustibles and hot work activities were controlled in accordance with plant procedures; (2) observed the condition of fire detection devices to verify they remained functional; (3) observed fire suppression systems to verify they remained functional and that access to manual actuators was unobstructed; (4) verified that fire extinguishers and hose stations were provided at their designated locations and that they were in a satisfactory condition; (5) verified that passive fire protection features (electrical raceway barriers, fire doors, fire dampers, steel fire proofing, penetration seals, and oil collection systems) were in a satisfactory material condition; (6) verified that adequate compensatory measures were established for degraded or inoperable fire protection features and that the compensatory measures were commensurate with the significance of the deficiency; and (7) reviewed the USAR to determine if the licensee identified and corrected fire protection problems.

- October 4, 2006, Containment Building 994' Level (Fire Area 30) (Please refer to

NRC Inspection Report 05000285/2006006. This sample is also being credited towards completion of inspection of the Nuclear Steam Supply System components during the Fall 2006 Refueling outage).

- October 16, 2006, Air Compressor and Auxiliary Feedwater area, Room 19 (Fire Area 32)
- October 16, 2006, Lower Cable Spreading area, Room 70 (Fire Area 41)
- October 23, 2006, Volume Control Tank area, Room 29 (Fire Area 20.3)
- November 7, 2006, Letdown Heat Exchanger Area III, Room 12 (Fire Area 6.7)
- November 11, 2006, Ion Exchanger area, Room 62, (Fire Area 20.5)

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed six samples.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

.1 Performance of Nondestructive Examination Activities Other than Steam Generator Tube Inspections

a. Inspection Scope

The inspectors observed the performance of three Class 1 welds and reviewed the welder certifications, welding procedures, welding procedure specifications, weld procedure qualification records and the final weld records for these welds.

The inspectors also reviewed the nondestructive examination associated with both replacement component installation and existing welds in service inspection activities, including: reviewing the radiographic film for three welds, observing dye penetrant examination of six welds, and observing ultrasonic examination of four welds. In conjunction with the observation and review of nondestructive examination activities, the inspectors reviewed procedures, reports, and technician qualification certifications.

The required sample size for this activity is one. The inspectors completed one sample.

b. Findings

No findings of significance were identified.

.2 Pressurized Water Reactor Vessel Upper Head Penetration Inspection Activities

The licensee replaced the reactor vessel upper head during this outage. Therefore, the inspectors did not perform this inspection step because the licensee did not perform any activities in this area.

The required sample size for this activity is one. The inspectors did not complete a sample because the licensee did not perform any activities in this area.

.3 Boric Acid Corrosion Control Inspection Activities

a. Inspection Scope

The inspectors reviewed the results of the boric acid walkdown, which was completed prior to onsite arrival. This included review of the procedures governing the walkdown and a review of a number of examination results, by reviewing both the tabulated results as well as pictures of boric acid deposits.

The required sample size for this activity is one. The inspectors completed one sample.

b. Findings

No findings of significance were identified.

.4 Steam Generator Tube Inspection Activities

Inspection Scope

The licensee replaced the steam generators during this outage. Therefore, the inspectors did not perform this inspection step because the licensee did not perform any activities in this area.

The required sample size for this activity is one. The inspectors did not complete a sample because the licensee did not perform any activities in this area.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed six inservice inspection related condition reports (CRs) issued during the current and past refueling outages, and verified that the licensee identified, evaluated, corrected, and trended problems. In this effort, the inspectors evaluated the effectiveness of the licensee's corrective action process, including the adequacy of the technical resolutions.

There is no required sample size for this activity.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

The inspectors observed testing and training of senior reactor operators and reactor operators to identify deficiencies and discrepancies in the training, to assess operator performance, and to assess the evaluator's critique. The training scenario involved a steam generator tube rupture observed on November 13, 2006. Documents reviewed by the inspectors included: Scenario SSG 84202b, "SGTR," Revision 1.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors reviewed the two maintenance activities listed below: (1) verify the appropriate handling of structure, system, and component (SSC) performance or condition problems; (2) verify the appropriate handling of degraded SSC functional performance; (3) evaluate the role of work practices and common cause problems; and (4) evaluate the handling of SSC issues reviewed under the requirements of the maintenance rule, 10 CFR Part 50 Appendix B, and the TSs.

- December 14, 2006, Containment Spray Injection Valve HCV-345 incorrect assembly
- December 21, 2006, Component Cooling Water Pump AC-3B unavailability due to repeated failures associated with the breaker's direct trip actuator

Documents reviewed by the inspectors included: CR 200604695 and 200603835, Maintenance Rule Functional Scoping Data Sheets for component cooling water and containment spray systems, and Maintenance Rule Cause Determination for Condition Report 200203680.

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

.1 Risk Assessment and Management of Risk

a. Inspection Scope

The inspectors reviewed the three assessment activities listed below to verify: (1) performance of risk assessments when required by 10 CFR 50.65 (a)(4) and licensee procedures prior to changes in-plant configuration for maintenance activities and plant operations; (2) the accuracy, adequacy, and completeness of the information considered in the risk assessment; (3) that the licensee recognizes, and/or enters as applicable, the appropriate licensee-established risk category according to the risk assessment results and licensee procedures; and (4) the licensee identified and corrected problems related to maintenance risk assessments.

- October 18, 2006, Toured the switchyard while work was performed with 345 Kilovolt electrical supply out of service and 161 Kilovolt supply feeding all critical loads
- December 14, 2006, Reviewed elevated risk condition with Diesel-Driven Auxiliary Feedwater Pump FW-54 out of service and associated compensatory actions
- December 20, 2006, Diesel Generator 1 monthly surveillance, troubleshooting the Auto-Standby feature of Component Cooling Water (CCW) Pump AC-3C, Condenser Off-Gas Radiation Monitor Replacement RM-057, and changing weather conditions

Documents reviewed by the inspectors included: Surveillance Procedure OP-ST-DG-0001, "Diesel Generator 1 Check," Revision 52 and the daily risk assessment profiles for the dates listed above.

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors: (1) reviewed plants status documents such as operator shift logs, emergent work documentation, deferred modifications, and standing orders to determine if an operability evaluation was warranted for degraded components; (2) referred to the USAR and design basis documents to review the technical adequacy of licensee operability evaluations; (3) evaluated compensatory measures associated with operability evaluations; (4) determined degraded component impact on any TSs; (5) used the Significance Determination Process to evaluate the risk significance of degraded or inoperable equipment; and (6) verified that the licensee has identified and implemented appropriate corrective actions associated with degraded components.

- July 18, 2006, found excessive regulator leakage on CCW nitrogen bottle Regulators NG-HCV-402A-R1, NG-HCV-400A-R1, NG-HCV-402B-R1, NG-HCV-403B-R1, and NG-HCV-401B-R1

- August 18, 2006, CCW Pump AC-3B Breaker 1B4A-1 repeatedly tripping free
- November 1, 2006, Containment Spray Injection Valve HCV-345 being incorrectly assembled in 2005 (This finding will be documented in NRC Inspection Report 05000285/2005018.)
- November 14, 2006, Reactor Coolant System flow instruments tubing separation
- November 22, 2006, Surveillance test failure of CH-143 (Boric Acid Pumps CH-4A & CH-4B Discharge to Charging Suction Header Check) and CH-155 (Charging Pumps CH-1A, B, & C Suction Header Gravity Feed Check Valve)

Documents reviewed by the inspectors are listed in the attachment. The inspectors completed five samples.

b. Findings

.1 Failure to Promptly Identify and Correct a Degraded Component Cooling Water Pump

Introduction. The inspectors identified a Green noncited violation of 10 CFR 50, Appendix B, Criterion XVI for the licensee's failure to promptly identify and correct a degraded component cooling water pump. The failure to recognize and fix this condition led to the pump being more likely to fail upon a valid demand to start.

Description. On June 23, 2006, electrical Breaker 1B4A-1 (Breaker Unit Component Cooling Water Pump AC-3B) failed to stay closed on two attempts to close it during routine maintenance. The breaker and the associated pump were subsequently declared operable and returned to service. The inspectors started reviewing this condition following an examination of the Condition Reporting system. The inspectors determined that two previous failures of the component cooling water Pump AC-3B to start occurred on May 24, 2004 and April 1, 2005. Further, the inspectors observed that a cause had not been determined for any of the three failures, nor had any (effective) corrective actions been taken. The inspectors questioned the licensee about potential causes and the extent-of-condition to components powered with General Electric AK-25 model breakers similar to that used for component cooling water Pump AC-3B. Subsequently, the pump failed to start on August 18, 2006 and September 7, 2006.

The licensee performed detailed analysis of the suspect breaker and determined that the cause for the spurious electrical trips (i.e., failures of the pump to start on a valid demand) was due to the lack of a notch in the reset paddle of the breaker. (The reset paddle is a fulcrum point that places tension on the spring that supplies mechanical force to drive the plunger when a tripping pulse is sent.) Without a notch in the reset paddle, the Direct Trip Actuator over traveled during a closing evolution and prevented the breaker from staying closed. The licensee performed extensive reviews, including visual examinations and high-speed video observations, to ensure that similar General Electric AK-25 model breakers installed in the plant were not subject to this condition. The licensee also determined that electrical breaker 1B4A-1 was degraded, but had been operable. The inspectors concurred with these assessments.

Analysis. The inspectors determined that the failure to promptly identify and correct a condition adverse to quality was a performance deficiency. This finding was determined to be greater than minor because the condition had an impact on availability/reliability of the component and thus affected the "Equipment Performance" attribute under the Mitigating Systems cornerstone. The inspectors evaluated this finding using Manual Chapter 0609, Appendix A, and determined that it was of very low safety significance (Green). This conclusion was reached because the finding was not a design or qualification deficiency, the finding did not represent a loss of safety function, was not an actual loss of safety function of a single train for greater than its TS Allowed Outage time, did not represent an actual loss of safety function for non-TS equipment, and was not potentially significant due to external events such as flooding, seismic occurrences, etc. This finding has a crosscutting aspect in the area of problem identification and resolution because the licensee failed to identify and correct the condition despite numerous opportunities to do so.

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion XVI, requires, in part, that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. Contrary to the above, the licensee did not take prompt corrective actions to correct the degraded component cooling water Pump AC-3B after identification of the problem on June 23, 2006, resulting in the pump being degraded. Since this failure to take prompt corrective action is of very low safety significance and was documented in the licensee's corrective action program, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000285/2006005-01). This violation was entered into the licensee's corrective action program as Condition Report 200603835.

.2 Failure to Determine Operability of Component Cooling Water Valves to Containment Cooling Units

Introduction. The inspectors identified a Green, noncited violation of TS 2.4. The violation was identified as a result of the licensee's failure to complete corrective actions two years ago that caused the licensee to incorrectly determine the operability of component cooling water inlet and outlet valves. These valves supply CCW to both the containment air-cooling and containment air-cooling/filtering units.

Description. Flow-induced hydrodynamic operation is a phenomenon in which water flow on the outside pipe bend, being of higher velocity, could cause an induced torque on a butterfly valve disc. This torque would force the valve to either open or close depending upon the valve orientation and valve proximity to the upstream bend.

The licensee has two containment air-cooling Units (VA-7C and VA-7D) and two containment air-cooling and filtering Units (VA-3A and VA-3B) as part of the system to control containment air temperatures during normal and accident conditions. The cooling coils for these units are cooled by the CCW system. The cooling coils for each unit can be isolated from the CCW system by two inlet and two outlet valves. Three of the four isolation valves for each cooling unit are air-operated butterfly valves with a safety-related nitrogen backup system to allow operation of the valves when the nonsafety-related air system fails. (This group of butterfly valves is referred to collectively as the HCV-400 series valves in this discussion.) The valves fail as-is on a

loss of air and nitrogen. Six of the twelve valves are subject to flow-induced hydrodynamic operation and will close on the loss of air and nitrogen thus securing CCW to the containment air cooling and containment air cooling/filtering units.

On June 29, 2006, the licensee reported in CRs 200602757 and 200602759 torn dust boots on two of the HCV-400 series valves. The initial operability determination concluded that the valves were operable because the licensee (incorrectly) decided a torn dust boot did not adversely affect the valves. Thirteen days later, the CCW system management system engineer reviewed the condition reports, inspected the valves and concluded the torn dust boots may have been indicative of leakage in the valve's air operator. The initial operability of one of the valves was changed to inoperable.

On July 18, 2006, the licensee reported in CR 200603019 leaks associated with the backup nitrogen supply regulators to the HCV-400 series valves. The operability determination for this condition concluded that the valves were operable since the design basis documents stated the air-operated valves failed as-is. The inspectors questioned this determination on why flow-induced hydrodynamic closure of the HCV-400 series valves was not considered. The inspectors also reminded the licensee that this same phenomenon was discussed approximately two years ago in NRC Inspection Report 05000285/2004003 and documented in CR 200401672.

While correcting the reported leakage, the licensee conducted further evaluation of the operability determination. The licensee concluded that the nitrogen leakage reported in CR 200603019 rendered the associated HCV-400 series valves inoperable. This conclusion was made after the valves were repaired and tested satisfactorily.

The licensee initiated CRs 200603765 and 200603808 to assess the conditions and ascertain the reportability of incorrect operability determinations on July 18 and June 29, 2006, respectively. The licensee determined these were reportable as TS violations. A root cause analysis determined that the events described above were caused by the failure to assign proper, timely corrective actions in 2004 to address the need for updating appropriate station documents used for determining operability of the HCV-400 series valves.

Analysis. The inspectors assessed this issue using the Significance Determination Process. The inspectors concluded that in 2004 the licensee failed to identify corrective actions involving the flow-induced hydrodynamic operation phenomenon of butterfly valves. This oversight resulted in violating TSs on June 29 and July 18, 2006, for the CCW inlet and outlet butterfly valves to the containment air-cooling units. This constitutes a performance deficiency since this was reasonably within the licensee's ability to foresee and correct. The finding was more than minor since it affected the "Containment Configuration Control" attribute of the Barrier Integrity cornerstone. Using Significance Determination Process, Manual Chapter 0609, the Phase One Analysis directs the use of Appendix H, since the finding involves the actual reduction in defense-in-depth for the atmospheric pressure control. Manual Chapter 0609, Appendix H, characterized the finding as having a very low safety significance because it was determined to have no impact on core damage frequency or large early release frequency.

The finding also has a crosscutting aspect in the problem identification and resolution area. The corrective action program component is affected because the licensee failed to take appropriate corrective actions to address the flow-induced hydrodynamic operation phenomenon of butterfly valves in a timely manner.

Enforcement. TS 2.4, "Containment Cooling," lists in (1)a.i. Containment Air Cooling and Filter Unit VA-3A and Containment Air Cooling Unit VA-7C. In (1)a.ii. the list contains Containment Air Cooling and Filtering Unit VA-3B and Containment Air Cooling Unit VA-7D. The TS reads in part,

"...b. During power operation one of the components listed in (1)a.i. and ii. may be inoperable. If the inoperable component is not restored to operability within seven days, the reactor shall be placed in hot shutdown condition within 12 hours. If the inoperable component is not restored to operability within an additional 48 hours, the reactor shall be placed in a cold shutdown condition within 24 hours. . ."

"(2) Modification of Minimum Requirements a. During power operation, the minimum requirements may be modified to allow a total of two of the components listed in (1)a.i. and ii. to be inoperable at any one time. . . If the operability of one of the two components is not restored within 24 hours, the reactor shall be placed in a hot shutdown condition within 12-hours. LCO 2.4(1)b. shall be applied if one of the inoperable components is restored within 24 hours. If the operability of both components is not restored within an additional 48 hours, the reactor shall be placed in a cold shutdown condition within 24 hours. . ."

Contrary to the above, the licensee violated TS 2.4(1)b on July 6, 2006, when a seven-day allowed outage time was exceeded and TS 2.0.1 on July 18, 2006, when a required shutdown was not completed. Specifically, on June 29 an HCV-400 series valve was incorrectly determined to be operable and the leakage was not corrected within seven days. On July 18, there were four HCV-400 series valves that were incorrectly determined operable, thus causing TS 2.4.(2) to be exceeded and requiring entry into TS 2.0.1. These TS violations are being treated as an NCV, consistent with the Section VI.A of the Enforcement Policy (NCV 05000285/2006005-02). This violation is in the licensee's corrective action program as Condition Report 200603808.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors selected the four postmaintenance test activities of risk significant systems or components listed below. For each item, the inspectors: (1) reviewed the applicable licensing basis and/or design-basis documents to determine the safety functions; (2) evaluated the safety functions that may have been affected by the maintenance activity; and (3) reviewed the test procedure to ensure it adequately tested the safety function that may have been affected. The inspectors either witnessed or reviewed test data to verify that acceptance criteria were met, plant impacts were evaluated, test equipment was calibrated, procedures were followed, jumpers were properly controlled, the test data results were complete and accurate, the test equipment

was removed, the system was properly re-aligned, and deficiencies during testing were documented. The inspectors also reviewed the USAR to determine if the licensee identified and corrected problems related to postmaintenance testing. (Please refer to NRC Inspection Report 05000285/2006006. All of the samples listed below are also being credited towards completion of inspection of the Nuclear Steam Supply System components during the Fall 2006 Refueling outage.)

- Inspection of the conduct of reactor coolant system (RCS) leakage testing and review of test results associated with replacement of the steam generators. Specifically, the inspectors observed the performance of Procedure OP-ST-RC-3007, "Periodic Reactor Coolant System Integrity Test," Revision 25, reviewed the selection of the test pressure, and observed the primary hydrostatic test of RCS components on November 25, 2006. Leakage was identified by the licensee on an incore instrument detector castle nut necessitating plant cooldown & repair.
- Inspection of the conduct of steam generator secondary side leakage testing and review test results. Specifically the inspectors observed the performance of procedures QC-ST-AFW-3002, "Auxiliary Feedwater Piping Forty-Month Inservice Test," Revision 3 and test QC-ST-MS-3001, "Main Steam Forty-Month Inservice Test," Revision 2.
- Inspection of the calibration and testing of instrumentation affected by steam generator replacement. Equipment included in the scope of this inspection included, but was not limited to, steam generator level indication, main steam flow rate detectors, main feed flow rate detectors, etc. For example, on November 9, 2006, a tilt was identified on installed replacement steam Generator RC-2B as described in CR 200605202. The inspectors evaluated the condition to determine the possible effect, especially the potential impact to the steam generator level indications.
- Inspection of the conduct of reactor coolant system leakage testing and review the test results associated with replacement of the pressurizer. Specifically the inspectors observed the performance of Procedure OP-ST-RC-3007, "Periodic Reactor Coolant System Integrity Test," Revision 25, reviewed the selection of the test pressure, and observed the primary hydrostatic test of RCS components on November 25, 2006. Leakage was identified by the licensee on an incore instrument detector castle nut necessitating plant cooldown & repair.

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed four samples.

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope

The inspectors reviewed the following risk significant refueling items or outage activities to verify defense in depth commensurate with the outage risk control plan, compliance with the TSs, and adherence to commitments in response to Generic Letter 88-17, "Loss of Decay Heat Removal:" (1) the risk control plan; (2) tagging/clearance activities; (3) reactor coolant system instrumentation; (4) electrical power; (5) decay heat removal; (6) spent fuel pool cooling; (7) inventory control; (8) reactivity control; (9) containment closure; (10) reduced inventory or midloop conditions; (11) refueling activities; (12) heat-up and cool-down activities; (13) restart activities; and (14) licensee identification and implementation of appropriate corrective actions associated with refueling and outage activities. The inspectors' containment inspections included observations of the containment sump for damage and debris; and supports, braces, and snubbers for evidence of excessive stress, water hammer, or aging. Documents reviewed by the inspectors are listed in the attachment. (Please refer to NRC Inspection Report 05000285/2006006. This sample is also being credited towards completion of inspection of the Nuclear Steam Supply System components during the Fall 2006 Refueling outage.)

The inspectors completed one sample.

b. Findings

Introduction. A Green self-revealing finding was identified for failure of operators to follow a standing operational procedure as required by TS 5.8.1.a. This failure resulted in less than the minimum number of raw water pumps required for decay heat removal from the spent fuel pool.

Description. On October 4, 2006, the licensee prepared to rotate raw water pumps and isolate the 'B' cell of the intake structure to support maintenance. The plant conditions were that the core was fully off-loaded to the spent fuel pool and cooling was provided by raw water Pumps AC-10C and AC-10D, which were operating in the 'B' and 'C' intake cells respectively. At 12:25 a.m., Raw Water Pump AC-10A was started and Raw Water Pump AC-10C was secured. At 2:16 a.m., the circulating water pump interconnection sluice Gates CW-16A and CW-16B were closed to support work on the 'B' intake bay cell. Shortly after this, alarms were received in the control room for elevated screen differential pressures, but the operators believed the alarm to be expected and only associated with lowering levels on the 'B' cell. At 2:26 a.m. fluctuating electrical current indications on the operating raw water Pump AC-10A caused operators to enter abnormal operating Procedure AOP-18, "Loss of Raw Water," Revision 6. At 2:31 a.m., the traveling screen sluice Gates CW-14A and CW-14B for 'A' intake cell were found closed. The event was terminated when operators secured raw water Pump AC-10A and opened sluice gates CW-14A and CW-14B to restore intake bay level.

During this transient, the operating raw water Pump AC-10A pumped down the isolated 'A' intake cell. Licensee Procedure SO-O-21, "Shutdown Operations Protection Plan," Revision 25, Attachment 7.2 required that two raw water pumps be available at all times for the plant conditions described above. With both the 'A' and 'B' intake cells isolated, three of the four raw water pumps were unavailable and only raw water Pump AC-10D

was supplying cooling water to the spent fuel pool. Further, this condition placed the plant in a "red" risk condition, which was prohibited by station procedures.

Analysis. The inspectors determined that the failure to follow the standing operational procedure was within the licensee's ability to control and hence was a performance deficiency. This finding was determined to be greater than minor in that it affected the "Configuration Control" attribute of the Mitigating Systems cornerstone, specifically "Shutdown Equipment Alignment." The inspectors attempted to use Manual Chapter 0609, Appendix G, because the condition occurred during shutdown conditions. The inspectors were unable to do so because an assumption contained in the worksheets was that fuel was in the reactor vessel. During this transient, all fuel was located in the spent fuel pool. Regional management determined that the finding was of very low safety significance (Green). The finding was evaluated considering Manual Chapter 0609, Appendix G, as a bounding case and was used as guidance to determine the significance of the finding. This finding has a crosscutting aspect in the area of human performance associated with work practices because the operator failed to use error prevention techniques like self-checking and peer checking, which would have prevented this event.

Enforcement. TS 5.8.1.a requires, in part, that written procedures be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, and Appendix A, 1978. Regulatory Guide 1.33, Revision 2, Appendix A, requires, in part, written procedures for Refueling and Core Alterations. Procedure SO-O-21, "Shutdown Operations Protection Plan," Revision 25, Attachment 7.2, required two raw water pumps to be available to facilitate heat removal. Contrary to the above, during a transient on October 4, 2006, only one raw water pump was available for removal of decay heat from the spent fuel pool. This violation of TS 5.8.1.a is being treated as a noncited violation, consistent with Section VI.A of the Enforcement Policy (NCV 05000285/2006005-03). This violation was entered into the licensee's corrective action program as CR 200604505.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the USAR, procedure requirements, and TSs to ensure that the five surveillance activities listed below demonstrated that the SSC's tested were capable of performing their intended safety functions. The inspectors either witnessed or reviewed test data to verify that the following significant surveillance test attributes were adequate: (1) preconditioning; (2) evaluation of testing impact on the plant; (3) acceptance criteria; (4) test equipment; (5) procedures; (6) jumper/lifted lead controls; (7) test data; (8) testing frequency and method demonstrated TS operability; (9) test equipment removal; (10) restoration of plant systems; (11) fulfillment of ASME Code requirements; (12) updating of performance indicator data; (13) engineering evaluations, root causes, and bases for returning tested SSCs not meeting the test acceptance criteria were correct; (14) reference setting data; and (15) annunciators and alarms set points. The inspectors also verified that the licensee identified and implemented any needed corrective actions associated with the surveillance testing.

- September 9 through December 31, 2006, Review of the post installation inspections, verification program and implementation for the steam generator replacement. (Please refer to NRC Inspection Report 05000285/2006006. This sample is also being credited towards completion of inspection of the Nuclear Steam Supply System components during the Fall 2006 Refueling outage.)
- October 13 and 29, 2006, Type C Local Leak Rate Test of Mechanical Penetrations Mike-39 and Mike-53
- November 22, 2006, In-office review of OP-ST-CW-3022, "AC-3C Component Cooling Water Pump Inservice Test," Revision 16.

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04)

a. Inspection Scope

The inspectors performed an in-office review of revisions to the Fort Calhoun Station Emergency Plan, including Revision 27 to Section B, and Revision 14 to Appendix C. The revisions were submitted in October 2006. The revisions relocated one field monitoring team from the Technical Support Center to the Emergency Operations Facility and added clarification for use of the cross-reference to NUREG 0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1.

The revisions were compared to their previous revisions, to the criteria of NUREG-0654 and NEI 99-01, "Methodology for Development of Emergency Action Levels," Revision 2, and to the standards in 10 CFR 50.47(b) to determine if the revisions were adequately conducted following the requirements of 10 CFR 50.54(q). This review was not documented in a safety evaluation report and did not constitute approval of licensee changes; therefore, these revisions are subject to future inspection.

The inspectors completed one sample during the inspection.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety [OS]

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope

This area was inspected to assess the licensee's performance in implementing physical and administrative controls for airborne radioactivity areas, radiation areas, high radiation areas, and worker adherence to these controls. The inspectors used the requirements in 10 CFR Part 20, the TSs, and the licensee's procedures required by TSs as criteria for determining compliance. During the inspection, the inspectors interviewed the radiation protection manager, radiation protection supervisors, and radiation workers. The inspectors performed independent radiation dose rate measurements and reviewed the following items:

- Performance indicator events and associated documentation packages reported by the licensee in the Occupational Radiation Safety Cornerstone
- Controls (surveys, posting, and barricades) of three radiation, high radiation, or airborne radioactivity areas
- Radiation work permits, procedures, engineering controls, and air sampler locations
- Conformity of electronic personal dosimeter alarm set points with survey indications and plant policy; workers' knowledge of required actions when their electronic personal dosimeter noticeably malfunctions or alarms
- Barrier integrity and performance of engineering controls in two airborne radioactivity areas
- Adequacy of the licensee's internal dose assessment for any actual internal exposure greater than 50 millirem Committed Effective Dose Equivalent
- Self-assessments, audits, licensee event reports, and special reports related to the access control program since the last inspection
- Corrective action documents related to access controls
- Radiation work permit (or radiation exposure permit) briefings and worker instructions
- Adequacy of radiological controls such as, required surveys, radiation protection job coverage, and contamination controls during job performance
- Posting and locking of entrances to all accessible high dose rate, high radiation areas, and very high radiation areas
- Radiation worker and radiation protection technician performance with respect to radiation protection work requirements

Either because the conditions did not exist or an event had not occurred, no opportunities

were available to review the following items:

- Licensee actions in cases of repetitive deficiencies or significant individual deficiencies

The inspectors completed 17 of the required 21 samples.

b. Findings

- .1 Introduction. The inspectors identified a self-revealing, NCV of TS 5.11.1, in which a worker failed to obtain a high radiation area (HRA) access authorization and associated radiological briefing prior to entering the posted area. This violation had very low safety significance.

Description. On October 24, 2006, a worker entered the containment building on radiation work Permit (RWP) 06-2542, "NSSSRP - Misc support," Task No. 1 for rigging and equipment moves. Electronic Alarming Dosimeter (EAD) alarm set points for this task were 25-millirem dose and 100-millirem per hour dose rates. This assignment did not require entry into a posted HRA. After entering the containment building and beginning work, the individual's foreman reassigned the person to a job on the temporary walkway above the reactor cavity. The individual should have changed to RWP 06-3538, which requires an HRA briefing from radiation protection prior to beginning work in the assigned area. This RWP would have also increased his EAD alarm set points to 100-millirem for dose and 150-millirem per hour for dose rate. The individual did not change RWP's and did not receive the HRA briefing prior to starting work in the new area. General area dose rates in the walkway were 60-80 millirem per hour. After working on the cavity walkway for a period of time, the individual's EAD alarmed at 25-millirem dose. The individual immediately exited containment and contacted radiation protection personnel.

Analysis. The failure to obtain an HRA access authorization and radiological briefing before entering the posted area is a performance deficiency. This finding is greater than minor because it is associated with one of the cornerstone attributes (exposure/contamination control) and affects the Occupational Radiation Safety cornerstone objective, in that the failure to obtain authorization for entry into the posted HRA and the radiological briefing could result in additional personnel exposure. Using the Occupational Radiation Safety Significance Determination Process, the inspectors determined that this finding was of very low safety significance because it did not involve: (1) an ALARA finding, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess doses. Additionally, this finding has a crosscutting aspect in the area of human performance work control because the foreman failed to appropriately coordinate work activities and evaluate the impact of changes to work assignments.

Enforcement. TS 5.11.1 states, in part, that in lieu of the "control device" required by 10 CFR 20.1601(a) and 20.1601c, each high radiation area, as defined in 10 CFR 20.1601, shall be barricaded and conspicuously posted as an HRA and entrance thereto controlled by a RWP. Any individuals permitted to enter such areas shall be provided with a continuously integrating and alarming radiation-monitoring device and may enter after the dose rate levels in the area have been established and personnel are made knowledgeable of them. Contrary to TSs, a worker entered HRA without obtaining the required radiological briefing and was not specifically authorized to enter the area.

Because this finding is of very low safety significance and has been entered into the licensee's corrective action program (CR 200604938), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000285/2006005-04, Failure to obtain HRA access authorization and associated radiological briefing.

- .2 Introduction. The inspectors identified a self-revealing, NCV of TS 5.11.1, in which a worker failed to wear an alarming device that could be heard while working in an HRA near the "A" steam generator cold legs. This violation had very low safety significance.

Description. On October 24, 2006, a contractor's ALARA Coordinator entered the containment building on RWP 06-3530, "Cut-out and weld-in of RCS piping to support replacement of steam generators." This area was a posted HRA with accessible areas where radiation exposure rates were greater than 100 millirem per hour. The individual intended to enter on Task No. 4, but inadvertently signed in on Task No. 1, which was suspended. The access control computer software will not prevent an individual from entering on a suspended task, but defaults to an EAD dose alarm of 1 millirem and a dose rate alarm of 1 millirem per hour. The alarm set points for Task No. 4 were 300 millirem for dose, and 2500 millirem per hour for dose rate. The individual entered the RCA and proceeded to the work location. The individual stated that the alarming dosimeter alarmed on "dose rate" shortly after entering the RCA but immediately cleared. The individual stated that they knew the alarm set points for Task No. 4 were much higher and that they had not entered any areas, which should cause the dosimeter to alarm. The individual believed the cause of the alarm to be a low battery. After requesting replacement of the battery, the individual entered the HRA. Due to the background noise level in the area, the individual was not able to hear the electronic dosimeter when it went into alarm at one millirem integrated dose. The individual worked in the area for a total of 1.7-hours. Upon exiting, the individual noticed the dosimeter alarm and immediately contacted radiation protection. The dosimeter indicated a total dose of 6-millirem.

Analysis. The failure to wear an alarming device that could be heard is a performance deficiency. This finding is greater than minor because it is associated with one of the cornerstone attributes (exposure/contamination control) and affects the Occupational Radiation Safety cornerstone objective, in that the failure to provide an adequate alarming dosimetry resulted in additional personnel exposure. Using the Occupational Radiation Safety Significance Determination Process, the inspectors determined that this finding was of very low safety significance because it did not involve: (1) an ALARA finding, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess doses. Additionally, this finding has a crosscutting aspect in the area of human performance work practices because the worker failed to use error prevention tools such as self- and peer-checking.

Enforcement. TS 5.11.1.b requires that an individual entering an HRA shall be provided with a radiation-monitoring device, which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. The fact that the background noise level was high enough that the worker could not hear the alarm effectively made the alarm nonfunctioning. Therefore, the failure to wear an alarming device that could be heard is a violation of TS 5.11.1.b. Because this finding is of very low safety significance and has been entered into the licensee's corrective action program (CR 200604938), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000285/2006005-05, Failure to

provide adequate alarming dosimetry.

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

The inspectors assessed licensee performance with respect to maintaining individual and collective radiation exposures ALARA. The inspectors used the requirements in 10 CFR Part 20 and the licensee's procedures required by TSs as criteria for determining compliance. The inspectors interviewed licensee personnel and reviewed:

- Current 3-year rolling average collective exposure
- Two outage or on-line maintenance work activities scheduled during the inspection period and associated work activity exposure estimates which were likely to result in the highest personnel collective exposures
- Site specific trends in collective exposures, plant historical data, and source-term measurements
- Site specific ALARA procedures
- ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements
- Interfaces between operations, radiation protection, maintenance, maintenance planning, scheduling and engineering groups
- Integration of ALARA requirements into work procedure and radiation work permit (or radiation exposure permit) documents
- Person-hour estimates provided by maintenance planning and other groups to the radiation protection group with the actual work activity time requirements
- Dose rate reduction activities in work planning
- Assumptions and basis for the current annual collective exposure estimate, the methodology for estimating work activity exposures, the intended dose outcome, and the accuracy of dose rate and man-hour estimates
- Method for adjusting exposure estimates, or re-planning work, when unexpected changes in scope or emergent work were encountered
- Exposure tracking system
- Use of engineering controls to achieve dose reductions and dose reduction benefits afforded by shielding
- Workers use of the low dose waiting areas
- First-line job supervisors' contribution to ensuring work activities are conducted in a dose efficient manner
- Records detailing the historical trends and current status of tracked plant source

terms and contingency plans for expected changes in the source term due to changes in plant fuel performance issues or changes in plant primary chemistry

- Source-term control strategy or justifications for not pursuing such exposure reduction initiatives
- Specific sources identified by the licensee for exposure reduction actions and priorities established for these actions, and results achieved against since the last refueling cycle
- Radiation worker and radiation protection technician performance during work activities in radiation areas, airborne radioactivity areas, or high radiation areas
- Self-assessments, audits, and special reports related to the ALARA program since the last inspection
- Resolution through the corrective action process of problems identified through postjob reviews and postoutage ALARA report critiques
- Corrective action documents related to the ALARA program and followup activities such as initial problem identification, characterization, and tracking
- Effectiveness of self-assessment activities with respect to identifying and addressing repetitive deficiencies or significant individual deficiencies

The inspectors completed 12 of the required 15 samples and 11 of the optional samples.

b. Findings

Introduction. The inspectors identified a self-revealing, NCV of TS 5.8.1.a, in which instructions for the use of a high efficiency particulate air (HEPA) filtration unit were not adequately incorporated into RWP instructions resulting in the contamination of three workers. The violation had very low safety significance (Green).

Description. On September 28, 2006, three individuals who had been cutting pressurizer instrument taps from the bottom of the pressurizer alarmed the personnel contamination monitors when attempting to exit the RCA. The individuals were logged into the RCA using RWP 06-3537 for pressurizer replacement activities including removal and installation of instrument lines. The work area was set up using scaffolding, with a small work platform, to access the bottom of the pressurizer. An HEPA ventilation unit was placed on the floor below the work platform with ductwork extending to the work platform and to an area near the instrument lines. The workers were given a briefing on dosimetry, dress requirements, and dose rates just prior to the start of the job. The removal of the instrument lines was performed using a portable band saw. Since the licensee had a history of failed fuel in previous fuel cycles, there was a high potential for highly contaminated residue to be inside the instrument lines. The HEPA ventilation duct was positioned on the work platform at the start of the cutting evolution, but was never placed near the cut locations on the instrument lines. During the cutting evolution, the band saw spread contamination over a large portion of the area directly beneath the pressurizer instrument taps, including the three workers on the platform. Upon exiting the work area, the three individuals alarmed the personnel contamination monitors. The individuals were decontaminated, and whole body counts were performed. Based on

whole body count results, the three individuals were assigned doses of 60, 75, and 86 millirem committed effective dose equivalent respectively.

Analysis. The failure to provide adequate work instructions is a performance deficiency. This finding is greater than minor because it is associated with one of the cornerstone attributes (exposure/contamination control) and affects the Occupational Radiation Safety cornerstone objective, in that the failure to incorporate adequate work instructions in the radiation work permit resulted in additional personnel exposure. Using the Occupational Radiation Safety Significance Determination Process, the inspectors determined that this finding was of very low safety significance because it did not involve: (1) an ALARA finding, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess doses. Additionally, this finding has a crosscutting aspect in the area of human performance resources because the licensee failed to provide complete and accurate work instructions in the RWP.

Enforcement. TS 5.8.1.a requires that procedures listed in Regulatory Guide 1.33, Appendix A, be established, implemented, and maintained. Section 7e. lists procedures for access control to radiation areas including a radiation work permit system. Contrary to the above requirements, the RWP instructions for the work activities did not contain adequate instructions on use and placement of the HEPA ventilation unit and ductwork. The "Worker Instructions" and the "Special Instructions" sections of the RWP did not address use of the HEPA ventilation unit. The ALARA Work Control Plan for RWP 06-3537 states, "A HEPA ventilation unit will be used during weld Prep." and, "A HEPA vacuum will be used for housekeeping on the work platform." Because this finding is of very low safety significance and has been entered into the licensee's corrective action program (CR 200604400), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000285/2006005-06, Failure to provide adequate instructions.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification (71151)

Occupational Radiation Safety Cornerstone

a. Inspection Scope

Occupational Exposure Control Effectiveness

The inspectors reviewed licensee documents from August 1, through November 17, 2006. The review included corrective action documentation that identified occurrences in locked high radiation areas (as defined in the licensee's TSs), very high radiation areas (as defined in 10 CFR 20.1003), and unplanned personnel exposures (as defined in NEI 99-02). Additional records reviewed included ALARA records and whole body counts of selected individual exposures. The inspectors interviewed licensee personnel that were accountable for collecting and evaluating these performance indicator data. In addition, the inspectors toured plant areas to verify that high radiation, locked high radiation, and very high radiation areas were properly controlled. Performance indicator definitions and guidance contained in NEI 99-02, "Regulatory Assessment Indicator Guideline," Revision 4, were used to verify the basis in reporting for each data element.

The inspectors completed the required sample (one) in this cornerstone.

Public Radiation Safety Cornerstone

- Radiological Effluent Technical Specification/Offsite Dose Calculation Manual
Radiological Effluent Occurrences

The inspectors reviewed licensee documents from August 1, through November 17, 2006. Licensee records reviewed included corrective action documentation that identified occurrences for liquid or gaseous effluent releases that exceeded PI thresholds and those reported to the NRC. The inspectors interviewed licensee personnel that were accountable for collecting and evaluating the PI data. PI definitions and guidance contained in NEI 99-02, "Regulatory Assessment Indicator Guideline," Revision 4, were used to verify the basis in reporting for each data element.

The inspectors completed the required sample (one) in this cornerstone.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

The inspectors evaluated the effectiveness of the licensee's problem identification and resolution process with respect to the following inspection areas:

- Access Control to Radiologically Significant Areas (Section 2OS1)
- ALARA Planning and Controls (Section 2OS2)

b. Findings.

No findings of significance were identified.

.2 Semiannual Trend Review

a. Inspection Scope

The inspectors performed a semiannual assessment (one inspection sample) of the licensee's corrective action program. The assessment covered condition reports written on CCW pump breaker failures. (Please refer to Section 1R15.b.1 of this report.) The inspectors specifically reviewed extent of condition concerns for General Electric AK-25 breaker failures. The inspectors reviewed the conditions found against operational experience from other plants as well as General Electric bulletins and notices.

b. Findings and Observations

No findings of significance were identified.

.3 Crosscutting Issue Aspects

The inspectors identified two findings with problem identification and resolution crosscutting aspects. One related to a degraded CCW pump was identified in Section 1R15.b.1; and a second was documented in Section 1R15.b.2, related to the failure to take appropriate corrective actions for hydrodynamic torque on butterfly valves.

4OA3 Event Follow-up (71153)

.1 (Closed) LER 05000285/2006003-00, Technical Specification Violation of Containment Air Coolers Due to Untimely Corrective Actions

The details of this condition are discussed in Section 1R15 of this report. This LER is closed.

.2 (Closed) LER 05000285/2006004-00, Loss of Shutdown Cooling Redundant Train Due to Valve Mispositioning

On September 9, 2006, the licensee commenced shutdown of the plant in support of the Fall 2006 refueling outage. On September 10, at approximately 9:30 a.m., operations personnel performed the initial valve lineup per Procedure OI-SC-1, "Shutdown Cooling Initiation," Revision 42, for establishment of shutdown cooling. (This procedure established the configuration of systems necessary to further lower plant temperature and maintain core cooling.) At 12:30 p.m., reactor coolant temperature decreased to less than 210°F and pressure was lowered below the necessary minimum for single reactor coolant pump operation. Once this condition existed, TS 2.1.1.(3) became applicable, and the steam generators became unavailable as a heat removal source due to inability to run reactor coolant pumps to dissipate decay heat. On September 12, at approximately 7:30 p.m., a valve lineup was subsequently performed for the purpose of re-verifying the configuration of the system. Operators performing this valve lineup discovered that manual isolation Valve SI-173 (Shutdown Heat Exchanger AC-4A & 4B Outlet Cross Connect Valve) was locked shut. The valve was immediately restored to the open position. The inspectors determined that had a failure of the operating 'A' train of Shutdown Cooling occurred, the 'B' train would not have been available. This issue was dispositioned in NRC Inspection Report 05000285/2006004, Section 1R20. This LER is closed.

.3 Inadvertent Over-Pressurization of Piping During Testing

a. Inspection Scope

The inspectors reviewed control room response to an unexpected pressurization event that occurred while conducting testing on November 17, 2006. As part of the follow-up to the inspectors observed plant chart recorders, reviewed control room logs, and discussed the event with Plant Management.

b. Findings

Introduction. A Green self-revealing finding was identified for failure to follow procedures during testing. This condition resulted in the damage to safety-related equipment and potential over-pressurization of chemical and volume control system (CVCS) and high-pressure safety injection (HPSI) piping.

Description. On November 17, 2006, the licensee was conducting testing of the Chemical and Volume Control and Safety Injection systems using OP-ST-CH-3006, "Chemical and Volume Control System (CVCS) and Safety Injection (SI) System Category C Valve Exercise Test," Revision 14. (The purpose of the test was to verify proper operation of check valves in the system.) Upon starting all three charging pumps per Step 6 of Attachment 1, noise was heard in the control room and all three charging pumps were stopped. Shortly thereafter, the Containment Coordinator reported to the Control Room that the packing leak-off line for High-Pressure Safety Injection to Reactor Coolant Loop 2A isolation Valve, HCV-318, had separated from its fitting and was leaking approximately 20-25 gallons/minute.

Attachment 1, Checklist A to the procedure required that the High-Pressure Safety Injection to Reactor Coolant Loop 2B Isolation Valve HCV-321 be open. The valve, which should have provided a discharge path for the pumps, was found in the shut position. During this transient, reactor coolant system inventory control function was being met by two of the HPSI Pumps SI-2A and SI-2B and the high-pressure safety injection alternate header isolation Valve HCV-2987. In response to the transient and its potential affect on HCV-2987 and other associate piping, the licensee swapped the reactor coolant system inventory control function to the charging pumps. The licensee performed system walkdowns of all components (joint connections, valves, caps, plugs, supports, hangers and braces), which were subject to the pressure transient and identified no other damaged components. Further, the licensee compared pressure data from the event and verified that system design pressures had not been exceeded.

Analysis. The inspectors determined that the failure to comply with required procedures was a performance deficiency. This finding was determined to be greater than minor in that it affected the "Configuration Control" attribute of the Mitigating Systems cornerstone, specifically "Shutdown Equipment Alignment." The inspectors evaluated this finding using Manual Chapter 0609, Appendix G, because the condition occurred during shutdown conditions. Using Checklist 2, the inspectors determined that the finding screened as Green because the condition did not increase the likelihood that a loss of decay heat removal would occur. This finding has a crosscutting aspect in the area of human performance associated with work practices because the operator failed to use error prevention techniques like self-checking and peer checking, which would have prevented this event.

Enforcement. TS 5.8.1.a requires, in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, and Appendix A, 1978. Regulatory Guide 1.33, Appendix A, requires, in part, written procedures for operation of safety-related systems, including the Chemical and Volume Control System. Procedure OP-ST-CH-3006, "Chemical and Volume Control (CVCS) and Safety Injection (SI) System Category C Valve Exercise Test," Revision 14 requires that HCV-321 Loop 2B HPSI Injection Valve be positioned open. Contrary to the above, on November 17, 2006, during testing control room operators failed to properly position the valve, which caused damage to equipment and the potential damage to other CVCS and HPSI piping. This violation of TS 5.8.1.a is being treated as an NCV, consistent with Section VI.A of the Enforcement Policy (NCV 05000285/2006005-07). This violation was entered into the licensee's corrective action program as CR 200605430.

.4 (Closed) LER 05000285/2006006-00, Inadvertent Start of Emergency Diesel Generator 2

On November 8, 2006, the plant was in a refueling outage with the core offloaded to the

spent fuel pool. At 11:40 a.m. emergency diesel generator DG-2 inadvertently started while de-energizing a vital 4160 VAC bus during a planned test. The diesel generator did not load onto the bus because the output breaker was in the pull-to-lock position for the test. No other safety-related systems were actuated. The LER was reviewed by the inspectors and no findings of significance were identified and no violation of NRC requirements occurred. The licensee documented the event in CR 200605235. This LER is closed.

4OA5 Other Activities

.1 (Closed) Temporary Instruction 2515/169, Mitigating Systems Performance Index (MSPI) Verification

a. Inspection Scope

During this inspection period, the inspectors completed a review of the licensee's implementation of the MSPI in accordance with the guidance provided in Temporary Instruction 2515/169. The review examined the licensee's implementation Document, "MSPI Basis Document," Revision 0, and verified the established system boundaries and monitored components were consistent with guidance provided in NEI 99-02, "Reactor Oversight Process Performance Indicators," Revision 4. The inspectors verified that the licensee did not include credit for unavailability hours for "short term unavailability" or "operator recovery actions to restore the risk-significant function" as is allowed by NEI 99-02.

Additionally, the inspectors reviewed the baseline MSPI unavailability time using plant specific values for the period of 2002 to 2004. The verification included all planned and unplanned unavailability. The plant specific data for 2005 to 2006 was also reviewed to ensure the licensee properly accounted for the actual unavailability hours of MSPI systems. For the same period, the MSPI component unreliability data was examined to ensure the licensee identified all failures of monitored components. The accuracy and completeness of the reported unavailability and unreliability data was verified by reviewing operating logs, condition reports, and work order documents. The unavailability and unreliability data was compared with performance indicator data submitted to the NRC to ensure that any discrepancies would not result in a change to the index color.

b. Findings

No findings of significance were identified. This completes the inspection requirements for this TI.

4OA6 Meetings

Exit Meeting Summary

On November 17, 2006, the inspectors presented the occupational radiation safety inspection results to Mr. J. Reinhart, Site Director, and other members of his staff who acknowledged the findings. Additional information concerning one of the violations was received after the exit meeting, resulting in the re-characterization of the finding. On

December 8, 2006, a final exit meeting was conducted via telephone with Mr. G. Cavanaugh, Manager Regulatory Compliance. The licensee acknowledged the findings presented in the exit meetings. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

On November 1, 2006, the inspectors presented the results of the emergency plan change inspection to Mr. C. Simmons, Supervisor, Emergency Preparedness. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

On October 27, 2006, the inspectors presented the in service results to Mr. J. Herman, Manager of Engineering Programs, and other members of the staff who acknowledged the findings. All proprietary information was returned to the licensee.

The results of the resident inspector activities were presented to Mr. J. Reinhart, Site Director, and other members of licensee management on January 11, 2007. The inspectors confirmed that proprietary information examined during the inspection period was returned to the licensee. Licensee management acknowledged the inspection findings.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

D. Bannister, Plant Manager
G. Cavanaugh, Supervisor Regulatory Compliance
S. Cofaul, ALARA Technician, Radiation Protection
M. Cove, Manager, System Engineering
H. Faulhaber, Division Manager, Nuclear Engineering
M. Fern, Manager, Shift Operations
M. Frans, Assistant Plant Manager
R. Haug, Manager, Radiation Protection
J. Herman, Manager, Engineering Programs
D. Guinn, Licensing Engineer
P. Kellogg, ALARA Technician, Radiation Protection
D. Lakin, Manager, Corrective Action Program
T. Maine, Supervisor, Radiation Protection
E. Matski, Compliance
J. McBride, Senior Radiation Protection Technician
J. McManis, Manager, Licensing
T. Nellenbach, Manager, Operations
T. Pilmaier, Manager, Chemistry
J. Reinhart, Site Director
R. Reno, Control Room Supervisor
M. Sandhoefner, Shift Manager
C. Simmons, Supervisor, Emergency Preparedness
J. Spiker, Supervisor Nuclear Projects
D. Spires, Outage/Work Management
D. Trausch, Manager Quality Assurance/Control
R. Westcott, manager, Nuclear Projects
C. Williams, Supervisor Radiation Protection Operations

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000285/2006005-01	NCV	Failure to Promptly Identify and Correct a Degraded Component Cooling Water Pump (Section 1R15.b.1)
05000285/2006005-02	NCV	Failure to Determine Operability of Component Cooling Water Valves to Containment Cooling Units (Section 1R15.b.2)
05000285/2006005-03	NCV	Inadvertent Pump Down of Intake Bay resulting in Less Than Required Raw Water Pumps (Section 1R20)

05000285/2006005-04	NCV	Failure to Obtain High Radiation Area Briefing (Section 2OS1)
05000285/2006005-05	NCV	Failure to Provide Adequate Alarming Dosimetry (Section 2OS1)
05000285/2006005-06	NCV	Failure to Provide Adequate Instructions (Section 2OS2)
05000285/2006005-07	NCV	Inadvertent Over-Pressurization of Piping During Testing (Section 4OA3)

Closed

05000285/2006-003-00	LER	Technical Specifications Violation of containment Air coolers Due to Untimely Actions (Section 4OA3.1)
05000285/2006-004-00	LER	Loss of Shutdown Cooling Redundant Train Due to Valve Mispositioning (Section 4OA3.2)
05000285/2006-006-00	LER	Inadvertent start of Emergency Diesel Generator 2 (Section 4OA3.4)

LIST OF DOCUMENTS REVIEWED

Section 1RO4: Equipment Alignment

OI-SI-1, "Safety Injection System Normal Operation," Revision 101

Drawing E-223866-210-130, "Composite Flow Diagram Safety Injection and Containment Spray System P&ID," Revision 38

Drawing 11405—254 Sh 2, "Flow Diagram Condensate P&ID," Revision 34

Drawing 11405—253 Sh 4, "Flow Diagram Steam Generator Feedwater and Blowdown P&ID," Revision 34

Drawing 11405—252 Sh 1, "Flow Diagram Steam P&ID," Revision 98

Section 1RO5: Fire Protection

Standing Order SO-G-28, "Station Fire Plan," Revision 66

Standing Order SO-G-102, "Fire Protection Program," Revision 7

Abnormal Operating Procedure AOP-6, "Fire Emergency," Revision 17
USAR, Section 9.11, "Fire Protection Systems"

Surveillance Procedure SE-ST-FP-0005, "Fire Barrier and Penetration Seals Eighteen Month

Inspection," Revision 14 completed on 7/28/06

Condition Report 200605227

Section 1R08: In service Inspection Activities

Condition Reports

CR 200503021
CR 200503463
CR 200504805
CR 200600406
CR 200604283
CR 200604592

Miscellaneous Documents

ECDR No. 06-0012, "Indication on Liner Plate," Revision B

NCR No. 06-0026, "Liner Plate," Revision 0

RT-ASME-XI, "Bechtel Nondestructive Examination Standard Radiographic Examination,"
Revision 3.

Work Order Package 00216645, "Aux Bldg Side Visual Inspection - Boric Acid Degradation,"
Revision 1

FCS-06-010, Memo from Kurt Saltzman (Authorized Nuclear In service Inspector) to Paul Hamer
(Omaha Public Power District), "Washington Group International NDE Procedure Review,"
October 14, 2006

RFP # 00001034, "Certified Design Specification for Replacement Steam Generators,"
Section 3, Revision 2

RFP # 00001034, "Section 'HB' Technical Specification, Alloy 690 Tubing Specification,"
Revision 2.

Calculation Number: FC07178, "Liner Plate Acceptance Criteria," Revision 0

Section P2.24 of the Quality Assurance Data Package for the Replacement Steam Generators,
"Pre-Service Inspection (ECT); Section 2 Pre-Service Inspection (ECT), Volume 3, Associated
Documents," Revision 0.

RT-128, NDE Report - Radiographic Film Interpretation: Dwg FSK—0028, Weld F-8A,
October 22, 2006.

RT-081, NDE Report - Radiographic Film Interpretation: Dwg FSK—0056, Weld F-6A,
October 12, 2006.

RT-130, NDE Report - Radiographic Film Interpretation: Dwg FSK—0028, Weld F-5A,
October 22, 2006.

Welding Procedure Specification P8-T(RA), August 18, 2006

PQR No. 1041, "Welding Procedure Qualification Record for Procedure Specification P8-T(RA)," January 13, 1999.

Field Welding Checklist Bechtel Job 25036, "RCS SG A Hot Leg Weld No. F-5-A," October 24, 2006

Field Welding Checklist Bechtel Job 25036, "RCS SG A Cold Leg Weld No. F-8-A," October 24, 2006

Field Welding Checklist Bechtel Job 25036, "RCS SG A Cold Leg Weld No. F-20-A," October 24, 2006

Certifications

2 Level II PT NDE Technician
1 Level III PT NDE Technician
3 Welders certified to automatic GTAW welding
1 Level II UT NDE Technician

Procedures

OPPD-IWE-92-1	Visual Examination of Class MC Components and Their Integral Attachments	0
OPPD-VT-98-1	Visual Examination: VT-1	1
OPPD-UT-98-13	Manual Ultrasonic Examination of Vessel Welds not Greater than two inches Thickness	0
OPPD-UT-98-9	Ultrasonic Examination of Cast Austenitic Piping Welds	1
OPPD-UT-98-2	Manual Ultrasonic Examination of Austenitic Piping Welds	2
OPPD-UT-98-1	Manual Ultrasonic Examination of Ferritic Piping Welds	2
OPPD-VT-98-3	Visual Examination for Mechanical and Structural Condition of Components	1
OPPD-PT-98-1	Liquid Penetrant Examination - Solvent Removable, Visible Dye Technique	1
PDI-UT-2	PDI Generic Procedure for the Ultrasonic Examination of Austenitic Pipe Welds	C
PDI-UT-1	PDI Generic Procedure for the Ultrasonic Examination of Ferritic Pipe Welds	C

NDE Examinations observed

UT examinations: all were on the pressurizer spray line - all pre-service NDE

4-PSS-1/04B

4-PSS-1/04C
4-PSS-1/07A
4-PSS-1/07B

PT examinations: all pipe welds in the SI system except one valve body weld

2-CH-28/07
2-HPH-2.22/20
2-HPH-2.24/13
2-HPH-2.24/SI-198 (valve body)
2-HPH-1.24/12
2-HPH-1.24/13

VT examinations: all inspected via record review of BACC inspections

Components:
CH-4A, CH-4B, SI-157, SI-170, SI-171, SI168, HCV-2948, HCV-2958, HCV-2459, LCV-383-1

Class 1 welding observed

Dwg FSK—0028 Welds F-8-A, F-5-A, and F-20-A (all welding performed from the i.d. of the piping).

Section 1R15: Operability Evaluations

Condition Reports:

200400008	200401628	200401672	200401785	200401815	200401880
200401881	200602669	200602715	200602716	200602757	200602759
200602911	200602911	200603019	200603546	20060371	200603648
200603765	200603808	200603835	200604073	200604488	200605049
200605484	200605488				

Fort Calhoun Station Corrective Action Program Root Cause Analysis Report, "Incorrect Operability Determination Resulting In Technical Specification Violation, Condition Report: 200603808, 200603765," dated October 23, 2006

Surveillance Procedure OP-ST-CH-3006, "CVCS and SI System Category C Valve Exercise Test," Revision 14

Calibration Procedure SP-CP-08-480-1B3C-4C, "Calibration of the Protective Relays for 480-1B3C-4C Bus," Revision 13

Preventative Maintenance Procedure EM-PM-EX-0202, "GE Type AK-2A-25 and AK-7A-25 Circuit Breaker Inspection," Revision 23

NRC Information Notice 85-58, "Failure of a General Electric Type AK-2-25 Reactor Trip Breaker"

NRC Information Notice 88-54, "Failure of Circuit Breaker Following Installation of Amptector Direct Trip Attachment"

Westinghouse Technical Bulletin TB-04-6, "DTA Test Procedure," dated March 11, 2004

Westinghouse InfoGram IG-03-1, "Inability of a Breaker Mounted GE AK-25 Direct Trip Actuator to Reset," Revision 1, dated March 4, 2003

Instruction Manual for Power Circuit Breakers Types AK-2/2A-15, AK-2/3/2A/3A-25, and AKU-2/3/2A/3A-25

Work Order 00248162, "Inspect Breaker Cubicle for Interferences with Breaker"

USAR Section 8.5, "Electrical Systems - Initial Cable Installation Design Criteria"

System Training Manual, Volume 11, Control Rod Drive System

Technical Specification 2.4, "Containment Cooling"

Operability Determination for Condition Report 200605049

Section 1R19: Postmaintenance Testing

Work Order 254670-01 - Leak Check of Affected Areas on #5 Incore Instrument Grayloc

Modification Construction Approval EC-39412, "Tubing Separation Modification for Steam Generators"

Pre-operational Test, EC 31589-T016, "RSG Functional Test: Steam Generator Level Transients," Revision 1

Calculation 06Q4630-CAL-001, "Stress Evaluation of Fort Calhoun Containment Liner Considering Concrete Voids," dated October 27, 2006

Bechtel Nonconformance Reports 06-0051 and 06-0053

Technical Specification 2.1.7, "Pressurizer Operability"

Report of Concrete Cylinder Tests, dated October 16, 2006

Results of Liner Plate Gouge Repair Leak Chase Pressure Test, dated October 23, 2006

Procedure IC-ST-RC-0030, "Channel Calibration of Pressurization Safety Valve RC-141 Tailpipe," Revision 5

Procedure IC-ST-RC-0028, "Channel Calibration of Pressurizer Pressure, Loop D/P-102," Revision 14

Drawing "Liner Plate Leak Chase Channel Weld Map," Revision 3

Drawing 25036-C-030, "Temporary Construction Opening Tendon Sheathing Restoration Details," Revision 0

Drawing 25036-C-031, "Temporary Construction Opening Reinforcing Restoration Details," Revision 0

Drawing 25036-C-032, "Temporary Construction Opening Tendon Restoration," Revision 0

Post Modification Testing Package for Replacement Reactor Vessel Head Installation (EC 33153)

Post Modification Testing Package for Nuclear Steam Supply Replacement Project (EC 31589) - Master Test Plan

Post Modification Testing for Installation of Reactor Coolant System Piping and Tubing for the Nuclear Steam Supply Replacement Project (EC 33104)

Post Modification Testing for Installation of Instrumentation Piping and Tubing for the Pressurizer for the Nuclear Steam Supply Replacement Project (EC 33105)

Section 1R20: Refueling and Other Outage Activities

Condition Reports:

200604296 200604327 200604723

Operating Instruction OI-RC-2A, "RCS Fill and Drain Operations," Revision 53

Operating Instruction OI-ST-10, "Turbine Tests," Revision 42

Operating Instruction OI-SC-1, "Shutdown Cooling System," Revision 42

Operating Procedure OP-4, Attachment 2, "Power Reduction," Revision 33

Procedure RE-CPT-RX-0001, "Post Refueling Core Physics Testing and Power Ascension," Revision 38

Shutdown Safety Advisor's Log dated September 13, 2006

Technical Specifications, Definitions Section, page 5

Drawing D-4768, "Primary Plant Simplified Flowpath Diagram," Revision 5

Drawing 25036-C-008, "Buried Utilities Composite Plan," Revision 0

Abnormal Operating Procedure AOP-19, "Loss of Shutdown Cooling," Revision 12

Root Cause Analysis Report for CR 200603965

Procedure OI-ST-10, "Turbine Tests," Revision 42

Estimated Critical Position Worksheet dated December 2, 2006

NRC Generic Letter 81-07, "Control of Heavy Loads"

Section 1R22: Surveillance Testing

Technical Specification Amendment 238, Section 2.6, "Containment System"

Technical Specification Amendment 238, Section 3.5(4), "Containment Isolation Valve Leak Rate Tests (Type C Tests)"

USAR Section 4.5.6.5, Revision 11, "In-Service Inspection of ASME Code Class 1, Class 2, and Class 3 Components"

SO-G-23, EC 37731, "Surveillance Test Program"

ANSI/ANS-56.8-2002, "Containment System Leakage Testing Requirements"
11405M-10, Sheet 2, Revision 014, "Aux Coolant Component Cooling System Flow Diagram P&ID"

11405M-10, Sheet 3, Revision 018, "Aux Coolant Component Cooling System Flow Diagram P&ID"

11405M-40, Sheet 1, Revision 036, "Aux Coolant Component Cooling System Flow Diagram P&ID"

11405M-40, Sheet 3, Revision 014, "Aux Coolant Component Cooling System Flow Diagram P&ID"

IC-91, Revision 007, "Dravo Piping Isometric (Aux. Coolant)"

IC-92, Revision 007, "Dravo Piping Isometric (Aux. Coolant)"

IC-323, Revision 007, "Dravo Piping Isometric (Aux. Coolant)"

Procedure EC-39925, "Power Operated Relief Valve (PORV) PCV-102-1 Exercise Test"

Section 2OS1: Access Controls to Radiologically Significant Areas (71121.01)

Corrective Action Documents

200603848, 200603904, 200603923, 200604442, 200604938

Audits and Self-Assessments

06-QUA-034, Radiation Protection Operations
SA-06-02, Radiation Protection Program

Radiation Work Permits

06-3512, Head Work in RHRA
06-3502, Minor Maintenance in HRA's and RHRA's

Procedures

RPI-13, Radiological Posting Standards, Revision 2
RP-202, Radiological Surveys, Revision 28

RP-204, Radiological Area Controls, Revision 46
RP-306, Hot Spot Identification and Tracking, Revision 17
SO-G-101, Radiation Worker Practices, Revision 30

Section 2OS2: ALARA Planning and Controls (71121.02)

Corrective Action Documents

200604040, 200604198, 200604201, 200604400,

Audits and Self-Assessments

06-QUA-043, Radiation Work Permits-ALARA

Radiation Work Permits

06-2537, NSSSRP-Pressurizer Upgrades
06-3537, NSSSRP-Pressurizer Replacement
06-3530, NSSSRP-Steam Generator RCS Cutout/Weld-in

Procedures

RP-AD-300, ALARA Program, Revision 13
RP-AD-500, Respiratory Protection Program, Revision 7
RPI-15, Evaluating Source Term for Radiation Protection Issues, Revision 1
RP-301, ALARA Planning/RWP Development and Control, Revision 26
RP-303, ALARA Cost-Benefit Analysis, Revision 5
RPI-650, Internal Dosimetry Program, Revision 9
RPI-606, Special Dosimetry Issue, Control, and Use, Revision 11

Section 4OA1: Performance Indicator Verification (71151)

Procedures

NOD-QP-40 NRC Performance Indicator Program, Revision 2

Miscellaneous

2005 Abnormal Batch Liquid and Gaseous Release Summary
2005 Batch Liquid and Gaseous Release Summary
2005 Liquid Effluents Continuous Mode
Surveillance Report Numbers: 63(3)-0606 and 63(3)-1105

Section 4OA3: Event Follow-up (71153)

Condition Reports 200605235

LIST OF ACRONYMS

ALARA	as low as reasonable achievable
CCW	component cooling water
CFR	Code of Federal Regulations

CR	Condition Report
CVCS	chemical and volume control system
EAD	electronic alarm dosimeter
HEPA	high-efficiency particulate air
HPSI	high pressure safety injection
HRA	high radiation area
MSPI	mitigating systems performance index
NCV	noncited violation
NRC	Nuclear Regulatory Commission
RCS	reactor coolant system
RWP	radiation work permit
SI	safety injection
SSC	structure system component
TS	Technical Specification
USAR	Updated Safety Analysis Report
WO	work order